NON-PUBLIC?: N
ACCESSION #: 8809010247
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Nine Mile Point Unit 2 PAGE: 1 of 6

DOCKET NUMBER: 05000410

TITLE: Reactor Scram due to Average Power Range Monitor Upscale Trip Caused by Personnel Error
EVENT DATE: 06/28/88 LER #: 88-026-00 REPORT DATE: 08/26/88

OPERATING MODE: 2 POWER LEVEL: 009

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: Robert E. Jenkins, Assistant Supervisor Technical Support
TELEPHONE #: 315-349-4220

SUPPLEMENTAL REPORT EXPECTED: No

ABSTRACT: On June 28, 1988 at 2012 hours Nine Mile Point Unit 2 (NMP2) experienced an actuation of an Engineered Safety Feature, specifically, a reactor scram. At the time of the event, the plant was in a startup condition with the reactor mode switch in the "Startup/Hot Standby" Position. Reactor pressure and coolant temperature were approximately 850 pounds per square inch gauge (psig) and 527 degrees F, respectively. The reactor was operating at approximately 9% of rated thermal power.

The scram occurred due to an Average Power Range Monitor (APRM) upscale trip. The immediate cause of the scram has been attributed to flow "changes" in the reheat system and subsequent changes in reactor steam loads. The root cause of this event has been determined to be operator error. Secondary causes of this event have been determined to be procedural deficiency and training deficiency.

Following the reactor scram, the Technical Specification (TS) maximum allowable cooldown rate of 100 degrees F/hour was exceeded. All applicable action requirements were met.

Initial action was to enter into operating procedure N2-OP-101C, "Scram Recovery". Corrective action has been to modify procedure N2-OP-2, "Moisture Separator Reheater System", and to issue a Lessons Learned Transmittal to

the Operations Department, and issue a Training Modification Recommendation to the Training Department.

(End of Abstract)

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I. DESCRIPTION OF EVENT

On June 28, 1988 at 2012 hours Nine Mile Point Unit 2 (NMP2) experienced an actuation of an Engineered Safety Feature, specifically, a reactor scram. At the time of the event, the plant was in a startup condition with the reactor mode switch in the "Startup/Hot Standby" Position. Reactor pressure and coolant temperature were approximately 850 pounds per square inch gauge (psig) and 527 degrees F, respectively. The reactor was operating at approximately 9% of rated thermal power.

Operators misinterpreted a caution in Operating Procedure N2-OP-2, "Moisture Separator Reheater System" that requires the temperature difference between the two steam separator reheaters to be less than 50 degrees F. While commencing a reactor startup on June 28, 1988, Operations personnel discovered the temperature increase of moisture separator 2MSS-E1B was lagging 2MSS-E1A. In order to equalize the moisture separator temperatures, the operator throttled open the low load steam inlet valve 2MSS-PV29A and 29B (Attachment 1). This caused increased steam loads (and reactor power) at low reactor pressure.

Operations personnel failed to recognize and identify the cause of the higher than normal power level for the plant conditions and were not adequately sensitive to the effect of increased steam loads on power with the APRM scram setpoints reduced (less than or equal to 15% with the mode switch in Startup).

A flow change in the reheat system caused changes in the steam loads being supplied by the reactor. This caused the reactor level and pressure to decrease initially. During this time reactor power began to increase as a result of increased feedwater flow. In the final 20 second prior to the reactor scram feedwater flow, reactor level, and reactor pressure were all increasing simultaneously causing reactor power to increase to the Average Power Range Monitor (APRM) trip setpoint (less than or equal to 15% reactor power with the mode switch in startup). A reactor scram then occurred.

Following the reactor scram, the Technical Specification (TS) maximum allowable cooldown rate of 100 degrees F/hour was exceeded. All applicable

TS action requirements were met. The cause of the excessive cooldown was the additional steam loads present with 2MSS-PV29A/B open.

There were no components or systems which were inoperable and/or out of service which contributed to the event. No plant system or component failures resulted from the event.

II. CAUSE OF EVENT

The root cause of this event has been determined to be operator error. The secondary causes of this event have been determined to be a procedural deficiency and training deficiency.

Operating Procedure N2-OP-2, "Moisture Separator Reheater System" requires the temperature difference between the two steam separator reheaters to be less than 50 degrees F. While commencing a reactor startup on June 28, 1988 Operators discovered the temperature increase of moisture separator 2MSS-E1B was lagging 2MSS-E1A. The upstream blocking valves had been opened to warm the piping up to the high/low load valves. In order to equalize the temperatures, the operators pressurized 2MSS-E1A and B via the low load steam inlet valves 2MSS-PV29A and B (Attachment 1). Operators failed to recognize that this caution was only applicable when the turbine was in service, and that the caution did not apply under the existing plant condition (turbine shell warming).

In this configuration, (2MSS-PV29B open) reactor pressure will be influenced by any flow change in the reheat system. The exact cause of the flow change in the reheat system could not be determined.

The root cause of this event was operator error. The operations personnel should not have been opening the reheat steam valves, and should have been sensitive to the effect on power and the reduced margin that existed to the APRM scram setpoint.

Secondary causes of this event have been determined to be procedural deficiency and a training deficiency. A precaution in N2-OP-2 required that the temperature difference between the reheaters be maintained less than 50 degrees F. The Operations personnel failed to understand the correct meaning of this caution.

III. ANALYSIS OF EVENT

The reactor scram which occurred was a conservative event and posed no adverse safety consequences. This event did not in any way adversely affect any safety system nor the operators' ability to achieve safe shutdown.

Following the reactor scram, the Technical Specification (TS) maximum allowable cooldown rate of 100 degrees F/hour was exceeded. Specifically, the vessel cooldown rate averaged over the first hour was approximately 117 degrees F/hour at the top of the vessel and less than 160 degrees F/hour at the vessel bottom head. The rationale for limiting the vessel cooldown rate is to avoid excessive structural fatigue due to thermal shock.

The reactor vessel vendor has provided an Engineering Evaluation which confirms that the cumulative fatigue usage of the vessel and its internals has not been adversely affected.

IV. CORRECTIVE ACTIONS

Initial corrective action was to enter into operating procedure N2-OP-101C, "Scram Recovery".

Corrective actions include the following:

1. A Lessons Learned document discussing this event has been written and included in the Operations Department's Lessons Learned Book.

2. Operating Procedure N2-OP-2, Section E, "Startup", has been modified. Specifically this includes:

a. Assuring reheater steam supply valves are closed until the turbine has reached 15% to 19% load.

b. Utilizing bypass valves 2MSS-V395 and V396 to increase reheater temperature and pressure.

3. Activities were performed to check possible malfunctions and misalignments and therefore a potential source of disturbance.

- Verified proper moisture separator reheater valve line-up (major components only)

- Verified proper operation of high and low load steam valves (2MSS-PV28A/B and 2MSS-PV29A/B)

- Verified proper operation of 2HRS-PS108. 2HRS-PS108 provides a pressure input to the logic of 2DSR-LV68.

- Verified calibration of 2CRS-PT103. 2CRS-PT103 provides the pressure input which modulates 2MSS-PV28A/B and 2MSS-PV29A/B.

No abnormal equipment operations or calibration setpoints were noted.

4. A Training Modification Recommendation (TMR) TMR 02-88.183 will be issued to the Training Department to provide a simulator demonstration similar to this event. The demonstration will stress the effects changing reactor steam loads has on system stability during startup.

Note - The following corrective actions pertain to limiting the reactor cooldown rate.

5. Immediate operator action was to shut the Main Steam Isolation Valves (MSIV) to minimize reactor cooldown rate. The MSIV's were closed approximately 25 minutes after the scram when it was realized the maximum cooldown rate of 100 degrees F/hour could be exceeded.

6. Changes have been made to Operating Procedure N2-OP-101C providing additional guidance to Operations personnel if an excessive cooldown condition occurs.

7. Technical Support will investigate to determine the cause of the excessive cooldown rate.

8. The reactor vessel vendor has provided an Engineering Evaluation which confirms that the cumulative fatigue usage of the vessel and its internals has not been adversely affected.

9. This incident and its root cause will be discussed with all shifts.

V. ADDITIONAL INFORMATION

A. Identification of Components Referred to in this LER

IEEE 803 IEEE 805
Component EIIS Funct System ID

Reheaters RHTR SB
Reactor RCT N/A
Pressure Control Valves PCV SB

B. Component Failures - None

C. There have been no previous similar events.

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ATTACHMENT 1

FIGURE OMITTED - NOT KEYABLE (DIAGRAM)

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NM NIAGARA
MOHAWK NMP37812
NINE MILE POINT NULEAR STATION / P.O. BOX 32 LYCOMING, NEW YORK 13093/TELEPHONE (315) 343-2110

August 26, 1988

United States Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

RE: Docket No. 50-410
LER 88-26

Gentlemen:

In accordance with 10 CFR 50.73, we hereby submit the following Licensee Event Report:

LER 88-26 is being submitted in accordance with 10 CFR 50.73 (a) (2) (iv), "Any event or condition that resulted in manual or automatic actuation of any Engineered Safety Feature (ESF), including the Reactor Protection System (RPS)."

A 10CFR50.72 (b) (2) (ii) report was made at 2046 hours on June 28, 1988.

This report was completed in the format designated in NUREG-1022, Supplement 2, dated September 1985.

Niagara Mohawk acknowledges that this Licensee Event Report is submitted late. The root cause and corrective actions originally developed were inadequate and the Site Operations Review Committee required extensive rewrites of this report. These rewrites caused the delay. The resident inspector was notified on July 28, 1988 that this Licensing Event Report would be late.

Very truly yours,

/s/ J. L. WILLIS
J. L. Willis
General Superintendent
Nuclear Generation

JLW/JMT/mjd

Attachments

cc: Regional Administrator, Region 1
Sr. Resident Inspector, W. A. Cook

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